

THE ACTIVITIES OF THE RADIATION PROTECTION MEASUREMENTS LABORATORY

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The Radiation Protection Measurements Laboratory (RPML) of the Institute of Atomic Energy is responsible for handling all the problems concerning radiation protection at the Institute and in the vicinity of Nuclear Centre (NC) Świerk and National Repository of Radioactive Waste (NRRW), located at Różan.

The main tasks of RPML are:

- Radiation monitoring of the Świerk and Różan sites.
- Surveillance of radiation safety.
- Radioactive waste control.
- Radiation protection in emergency conditions.
- Improvement of radiation protection measurements and methods.
- Calibration of radiation protection monitoring instruments.
- Personal dosimetry.
- Sewage and drainage water activity measurements.
- Environmental radiation monitoring.

The following laboratories and facilities are available in the Laboratory:

- Mixed radiation fields laboratory (MRF).
- Whole body counter (WBC).
- Counter of thyroid activity (TC).
- Calibration laboratory with standard radiation sources (Calibration Division, CD).
- Environmental measurements laboratory.
- Radiochemical laboratory (RL).

On December 31, 2006 the Laboratory employed 18 graduate and 10 non-graduated staff.

In 2006 RPML continued successfully the activities concerning improvement of measurement procedures in two main domains of Laboratory accredited by the Polish Centre for Accreditation (PCA):

- The determination of internal body contamination (whole body, thyroid and excretions) – accreditation No. AB 567.
- Calibration of dosimetric instruments (gamma, neutron and surface contamination monitors) – accreditation No. AP 070.

The improved versions of Quality Manual as well as all the procedures were elaborated and some technical and organizational activities undertaken, especially: validation, traceability and estimation of uncertainties of the methods used.

Several internal quality audits were performed, opinions collected from experts, and in February 2006 the yearly PCA audit took place for AB 567 and in October 2006 for AP 070, both with positive results.

The research activities of Laboratory are described in the next part of Annual Report.

The technical activities of the Laboratory comprised in 2006:

- Whole body, thyroid and “in vitro” monitoring were carried out for radiation workers from NC Świerk and external customers 353 WBC, 154 TC and 159 RL measurements were made during 2006 (Tables 1-3). The results of measurements of ^{137}Cs internal activity in people, from 1986 to 2006, are presented in Fig. 1.
- Regular monitoring of radiation workers was carried out with thermoluminescent dose meters (TLD). Values of individual dose equivalents registered in 2006 are below the Annual Dose Limit.
- The environmental monitoring within and outside the NC and NRRW boundaries included the measurements of direct or stray radiation due to the operation of reactors, accelerators, etc. and the measurements of radioactivity in samples of air, rivers and underground water, soil, sediment, mud and vegetation. In 2006 more than 1000 environmental samples were measured (Tables 4-6).

Table 1. Whole Body Counter measurements.

Dose	Measurements	Persons
< 1% E _w	351	251*
> 1% E _w	2	2
Total	353	253

* - 26 persons were contaminated by ^{131}I including 8 persons measured by Thyroid Counter and calculated the committed dose equivalent.

E_w – limit of annual effective dose.

Table 2. Thyroid Counter measurements.

Dose	OBRI I-131		OTHERS I-131	
	Measurements	Persons	Measurements	Persons
< 1% E _w	127	18	-	-
> 1% E _w	27	6	-	-
Total	154	24	-	-

Table 3. Measurements of biological probes.

	Measurements	Persons	<1% E _w	>1% E _w
Total α-activity*	6	6	6	-
Total β-activity	120	86	86	-
Activity of P-32	5	3	3	-
Activity of S-35	5	3	3	-
Activity of tritium (HTO)	16	9	9	-
Activity of Sr-90	7	4	4	-

* - Detection limit for α activity is > 10% of E_w

The results of measurements show that there is no recorded influence of the NC and NRRW on the environment and the population living in the vicinity of both facilities.

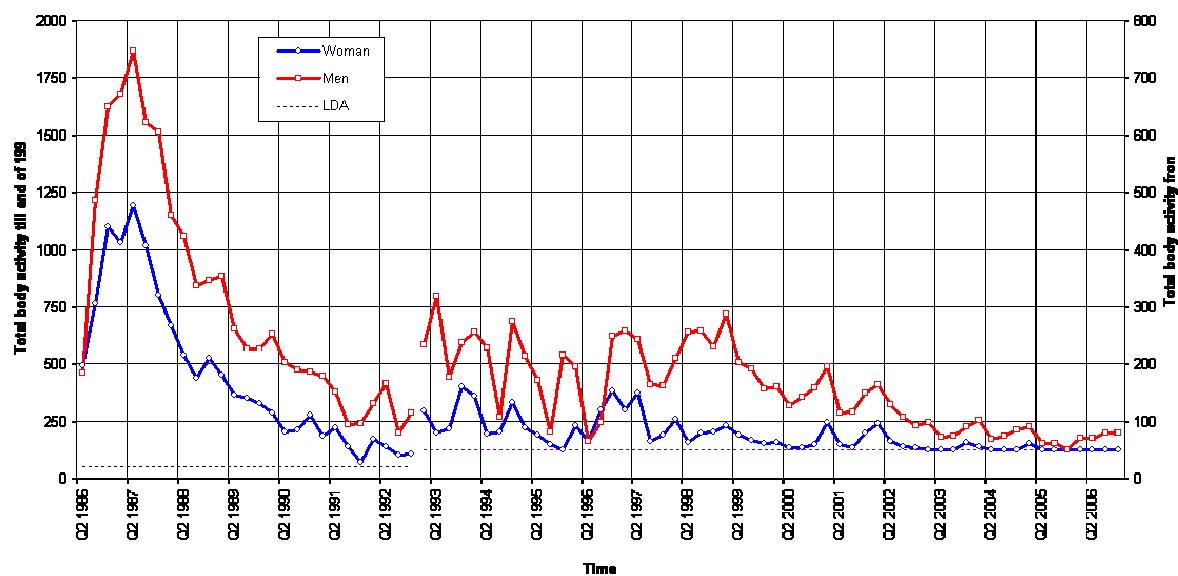


Fig. 1. The activity of ^{137}Cs in people after the Chernobyl accident (mean per person).

Table 4. Total activity of β -radiation in water inside NRRW Rózan.

Type of probe and place of collection	Probe	2005		2006	
		average	min	max	median
Ground waters inside NRRW Rózan	10p	<0.08	< 0.08	0.10	<0.09
	11p	0.09	< 0.08	0.25	0.16
	12p	0.18	0.21	0.35	0.29
	17p	0.10	0.10	0.16	0.12
	18p	<0.08	<0.08	0.09	0.18
Tap water (inside NRRW)	131	0.89	0.10	0.85	0.54
	F-R	<0.08	< 0.08	0.15	0.11

Table 5. Activity of tritium (HTO) in waters inside and outside NRRW Rózan.

Collection place	Probe	2005		2006		
		Average	I Q	II Q	III Q	IV Q
Ground waters inside of NRRW Rózan	10p	<4	< 4	< 4	< 4	< 4
	11p	330	370	330	300	260
	12p	3550	5500	1800	4400	3400
	17p	745	740	700	670	600
	18p	9	<4	<4	<4	< 4
	130p	185	190	180	180	180
	131p	18500	17000	15000	15000	16000
	132p	168	91	65	12	<4
Ground waters outside NRRW	F-1	48	<4	35	35	22
Tap water (inside NRRW)	F-R	<4	< 4	< 4	< 4	< 4

Table 6. Activity [Bq/kg] of environmental probes (soil and grass) inside and outside NRRW Różan in 2006.

Probe	Place	K-40	Cs-137	Ac-228 (Th-232)	Ra-226 (U-238)
Soil - II quarter					
<i>inside</i>	G 706	310	58	7	12
	707	530	140	13	18
<i>outside</i>	701	620	130	22	20
	703	340	7	121	13
Soil - III quarter					
<i>inside</i>	G 706	330	50	8	8
	707	550	110	17	21
<i>outside</i>	701	620	25	20	20
	703	480	3	16	15
Grass - II quarter					
<i>inside</i>	R 706	1100	3	<7	<3
	R 707	1200	1,9	<7	<3
<i>outside</i>	R 701	1100	20	<7	<3
	R 703	1800	1,3	<7	<3
Grass - III quarter					
<i>inside</i>	R 706	980	1,8	<7	<3
	R 707	980	5,2	<7	<3
<i>outside</i>	R 701	1200	29	<7	<3
	R 703	1200	7,2	<7	<3

The Calibration Laboratory is maintaining and using the standard fields of neutron and gamma radiation. The ^{137}Cs is the main calibration source of gamma radiation. Standard neutron fields, traceable to primary standard laboratory, which is the National Physical Laboratory (NPL, Great Britain) were established at the Institute of Atomic Energy almost ten years ago. The fields are formed by calibrated sources of ^{252}Cf and $^{241}\text{Am-Be}$. For routine use, there is also $^{239}\text{Pu-Be}$ neutron source available calibrated against standard source of $^{241}\text{Am-Be}$. Additionally, spherical filters made of iron or paraffin can be used for modification of the neutron spectrum and gamma component of absorbed dose.

The neutron fields are used mostly for research as well as the only fields in Poland suitable for calibration of neutron dose meters employed in radiation protection. Maintenance of the fields includes periodic assessment of their dosimetric parameters, development of measuring methods and international inter-laboratory comparisons. The parameters of neutron fields in the IAE calibration hall have been periodically checked. In 2006 the laboratory participated in International Neutron Intercomparison EUROMET Project No. 608. The results of the test should be revealed in 2007.

In the field of the ^{137}Cs gamma source the periodical test measurements were performed applying specially designed and constructed reference instrument. The Kerma rate in gamma radiation fields of the new gamma irradiator were measured by Central Office of Measures.

The complementary tests with own Laboratory reference instrument were initiated.

Calibration Laboratory performs calibrations of surface contamination monitors, using reference sources of beta and alpha radiation. In 2006 a new standard source ^{36}Cl for contamination monitors was purchased and implemented for routine use.

Total number of monitors calibrated in 2006:

- Surface contamination monitors - 133
- Gamma dose and dose rate monitors - 182
- Neutron dose equivalent monitors - 3

In 2006 the staff of CD performed also the periodical tests of dosimetric monitoring system of MARIA reactor as well as the calibration of data lines of the system with the detectors.

The calibration of radiation monitors in liquid radioactive waste storage tanks of Radioactive Waste Management Plant has been performed as well as calibration of radiation monitoring personal gate.

RISK MANAGEMENT AND RADIOLOGICAL SAFETY IN HEALTH CARE

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The health care sector is an extensive and complex sector. Most likely, it is the largest single employment sector in Europe, where a large range of very different tasks and jobs are carried out. It therefore contains a large number of hazards and risks and complex occupational, safety and health management problems.

The complex combination of processes, technologies and human interactions that constitutes the modern health care delivery system involves an inevitable risk of adverse events that can happen. Treatment errors in medicine are classified under various names including “treatment misadministrations”, “treatment incidents”, “treatment accidents”, “unusual occurrences”, “treatment discrepancies” and “adverse events”. In 2000, the Institute of Medicine in the United States published a detailed report on treatment errors in medicine, entitled “To Err is Human: Building a Safer Health System” [1]. It was estimated that there were about 44 000 to 98 000 people in the United States who died annually from medical errors. The estimated total annual cost of these errors was \$38 to \$50 billions per year. The most common types of errors were categorized as being related to “technical” (44%), “diagnosis” (17%), “failure to prevent injury” (12%), and “use of drugs” (10%).

Not only patients, but also all occupational groups in the medical sector may face hazards. The main risk factors are well documented [2]. They include musculoskeletal loads, biological agents, chemical substances, radiological hazards, changing shifts, work rhythms and night work, violence from members of the public, accidents at work, etc.

Additionally to the general safety systems there is a special system for radiological safety of medical devices and for radiation protection of patients and medical personnel. In this work [3], radiological protection is considered within the system approach, in order to underline, that risk assessment and effective health and safety management are the key to preventing and reducing patient and healthcare-worker exposure to any kind of hazard. This includes also the safety of medical devices and systematic assessment of direct and indirect consequences of particular technologies and threats against patient safety due to misuse, overuse or underuse of technology.

There are many models of complex systems and their malfunction but the most popularly in medicine is so called Swiss Cheese. The key terms of the model are “active errors” and “latent errors”. Active errors occur

at the point of contact between a human and some aspect of a larger system (e.g., a human-machine interface). They are generally readily apparent (e.g., pushing an incorrect button, ignoring a warning light) and almost always involve someone at the frontline. Latent errors (or latent conditions), in contrast, refer to less apparent failures of organization or design that contributed to the occurrence of errors or allowed them to cause harm to patients.

Concerns regarding patient safety, variable health care quality and increasing health care costs have led to the introduction of clinical management tools that have their origins outside of the traditional health care sector. The degree of regulation imposed on any device or practice is proportional to the potential hazard associated with this device or practice. This approach is known as risk management and it is based on one of the cardinal concepts in patient safety, borrowed from industry, which is systems analysis. This is the concept that systems failures – not individual human failures – are to blame for many of the adverse events occurring in health care.

The step further is Complexity Theory which emphasizes interactions between a local system and its environment (such as the larger system in which a given hospital or clinic operates). It is often tempting to ignore the larger environment as unchangeable and therefore outside the scope of quality improvement or patient safety activities. According to complexity theory, behaviour within a hospital or clinic (e.g., non-compliance with a national practice guideline) can often be understood only by identifying interactions between local attributes and environmental factors.

The application of safety culture concept used in other high hazard organizations, like in nuclear and aviation industries are being considered for health care.

References

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IMPROVED HIGH VOLTAGE SUPPLY FOR RECOMBINATION CHAMBERS

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High precision dosimetric measurements performed with recombination chambers require high stability of the voltage supplied. Tests [1,2] carried out on the electronic circuits used for high voltage (HV) supply to recombination chambers resulted in identification of their main parameters crucial for stability of the recombination chamber readings. To improve the stability of HV few changes in the design of the appropriate circuits have been proposed and tested [3].

Digitally controlled high voltage supply used for recombination chambers comprises the digital-analogue circuit U7, operational amplifier U13 and modular voltage converter U11 (Fig.1). Unfortunately, it turned out that that design does not provide sufficient stability of output voltage higher than 1000 V, because voltage converter warms up and output voltage falls about 600 mV during first hour after setting the desired voltage. To eliminate this effect the radiator was used on voltage converter.

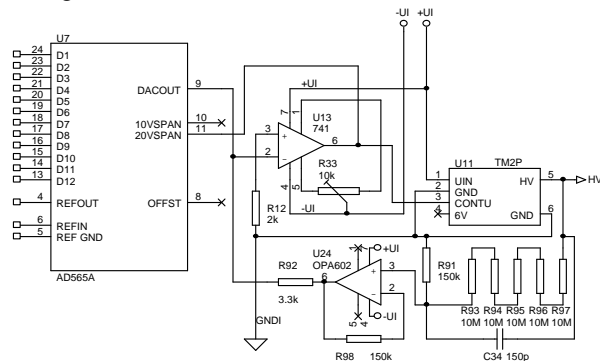


Fig. 1. Improved design of digital controlled high voltage supply.

As a further improvement a feedback from output high voltage supply to input amplifier U13 was incorpo-

rated. This feedback is realised as the voltage divider consisting of resistors R91 and R93-R97, voltage follower circuit U24 and resistor R92. Full feedback current that flows to pin #2 of the amplifier U13 depends now in 75 % on the HV output and in 25 % on the output of amplifier U13 itself (Fig.1).

Test consisting in recording the HV output over one hour after setting the supply to the nominal value of 1.2 kV revealed only the slow decrease by ~ 100 mV. The temperature of the voltage converter fluctuated within ± 0.3 C around 18.3 C.

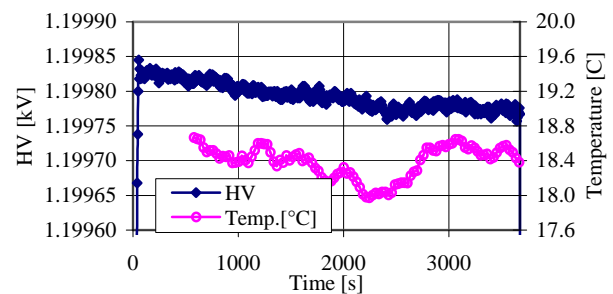


Fig. 2. High voltage output and temperature of voltage converter during first hour after setting 1.2 kV HV supply.

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- [1] Z. Rusinowski, IAE Report B-41 (1998)
- [2] Z. Rusinowski, IAE Report B-50 (1999)
- [3] Z. Rusinowski, IAE Report B-32 (2006)

A COMPARISON OF DIFFERENT RECOMBINATION METHODS IN MIXED RADIATION FIELDS AT HIGH ENERGY ACCELERATORS

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Recombination chambers have been successfully used for dosimetry of mixed radiation fields, especially near high-energy accelerators, for over 40 years. Several dozen recombination chamber types have been designed and a large number of recombination methods and measuring techniques have been developed. All of them are based on the phenomenon of initial recombination of ions in a gas in the chamber cavity, and the choice of a particular method depends usually on measuring conditions and on the requested type and accuracy of the results.

The aim of this work was to compare the values of the ambient dose equivalent rate, $H^*(10)$, determined by eleven different recombination methods, selected for their relative lack of complexity in practical use [1, 2]. Most of them were investigated earlier in a large number of different mixed radiation fields, covering the energy range from thermal to high energy neutrons, and in the radiation fields composed of different kinds of radiation. A few newer methods which are currently under investigation were also included in the measurements and the comparison.

For the purposes of this work, the ambient dose equivalent rate was expressed by one general formula:

$$H^*(10) = A^{-1} i_S Q_i g(Q_i),$$

for almost all the methods compared. $i_S = i(U_S)$ is the ionisation current measured at a polarising voltage U_S applied to the chamber. U_S should be high enough to ensure an ion collection efficiency of above 0.99 when the chamber is irradiated in a gamma radiation field.

$A^{-1} = [H^*(10) / i_S]_{Cs}$ is the calibration factor, i.e. the reciprocal of the chamber sensitivity to the reference gamma radiation (^{137}Cs gamma radiation source). Q_i is the index of radiation quality, specific for a particular recombination method. $g(Q_i) = 1 - \beta_i (Q_i - 1)$ is a correction factor which depends only on the measured value of the radiation quality index and substitutes for other corrections, e.g. for neutron energy. β_i is determined in a reference radiation field of mixed neutron + gamma radiation (in this work – from a ^{239}Pu -Be source), where the value of $H^*(10)_{ref}$ is known, as:

$$\beta_i = \left[\frac{1 - g(Q_i)}{Q_i - 1} \right]_{ref} \quad \text{where} \quad g(Q_i) = \frac{H^*(10)_{ref}}{A^{-1} i_S Q_i}.$$

All the recombination methods considered in this work [1, 2] resulted in $H^*(10)$ differing less than 20% (Table 1). Such uncertainty is usually acceptable in radiation protection. Therefore, simple recombination methods can be used in most situations. Our results have proved that the recombination chambers of the REM-2 type are especially suitable for monitoring mixed radiation fields in facilities employing high energy accelerators.

References

- [1] M. Zielczyński et al., in: Proc. Tenth Symposium on Neutron Dosimetry, Uppsala, Sweden, Book of Abstracts B1-5 (2006)
- [2] M. Zielczyński et al., IAE Report B-33 (2006) (in Polish)

Table 1. Comparison of the $H^(10)$ values obtained by different recombination methods [1, 2] in CERN-CERF radiation fields (top iron TI6 and side concrete SC2 positions) and at proton accelerator at INP Kraków behind the eye phantom.*

Method	(TI6)			(SC2)			(INP)		
	Q_i	g_i	$\frac{H^*(10)}{H_{ref}^*}$	Q_i	g_i	$\frac{H^*(10)}{H_{ref}^*}$	Q_i	g_i	$\frac{H^*(10)}{H_{ref}^*}$
1 4U	7.30	0.874	1.0	3.23	0.96	0.9	7.7	0.878	1.085
2 2U	8.87	0.877	1.14	3.48	0.95	0.96	8.0	0.846	1.078
3 U ⁿ	9.6	0.71	1.07	3.30	0.917	0.88	7.8	0.769	0.96
4 i _A dir	7.3	0.874	0.92	3.23	0.96	0.87	7.7	0.878	1.02
5 i _A ch	7.4	0.872	1.01	3.25	0.96	1.04	7.9	0.86	1.08
6 RMM		1	1.03		1	1.02	8.25	1	1.14
7 ERM	9.32	0.71	1.04	3.3	0.92	0.88	10.4	0.67	0.97
8 VIQ	11	0.67	0.91	4.5	0.90	1.09	13	0.58	1.04
9 H-press.							6.1	1.3	0.78
10 BF ₃									0.81
11 CO ₂							7.4	0.87	0.89

IONISATION CHAMBER CONTAINING BORON AS A NEUTRON DETECTOR IN MEDICAL ACCELERATOR FIELDS

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Medical electron accelerators of energy above 10 MeV usually generate also neutrons. These, so called photon-neutrons, arise due to photon-neutron nuclear reactions, mainly in the target and in the beam collimator. They are emitted nearly isotropically and create a neutron radiation field in the treatment room. The whole patient's body is irradiated by the photon-neutrons. Neutrons can reach also neighbouring rooms, if neutron-absorbing elements are not introduced to the shielding walls. Therefore, two quantities are of interest for radiation protection – the neutron component of the absorbed dose in the patient's organs, and the ambient dose equivalent of mixed radiation, $H^*(10)$.

In present studies, recently designed ionisation chambers containing boron, have been investigated in the radiation field at a 15 MV medical accelerator Varian Clinac 2300C/D. The main idea of the detector design was to combine the common principle of a thermal neutron detector inside a moderator with the features of recombination chambers, in order to design the detector with similar sensitivity to $H^*(10)$ of photons and neutrons at medical accelerators. Then, the signal of the detector would be proportional to $H^*(10)$ with acceptable dependence on composition (neutron to photon dose ratio) of the radiation field.

It is well known that introducing boron to an ionisation chamber increases its sensitivity to neutron radiation. High pressure ionisation chambers are especially advantageous because of the possible use of the initial ion recombination phenomenon to design the chambers with a desirable ratio of neutron to photon response. The idea of this work was to design a chamber whose sensitivity to neutrons in respect to absorbed dose is higher than the sensitivity to photons by a factor close to the neutron quality factor, i.e. the sensitivity in respect to the dose equivalent is the same for photons and neutrons.

Three types of ionisation chambers containing boron have been designed at the Institute of Atomic Energy [1, 2]. Two of them were filled with BF_3 up to the gas pressures between 300 and 500 kPa (gas density above 8 kg/m^3). At such gas density there is considerable initial recombination of ions in the gas cavity, especially at low polarizing voltages. Since this kind of recombination does not depend on the dose rate, so the chamber sensitivity can be determined as a function of the polarizing voltage and the appropriate operating

voltage can be chosen for optimisation of the chamber response. The response, i.e. the sensitivity in respect to $H^*(10)$ for the REM-2 chamber filled with BF_3 and operated at 400 V, is nearly the same for photons and for fast neutrons (Fig. 1).

The third chamber was a free air ionisation chamber with tissue-equivalent electrodes covered by B_4C .

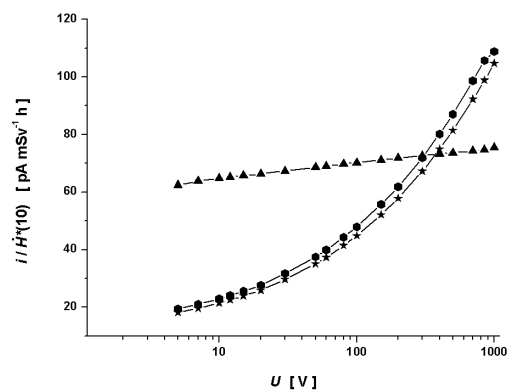


Fig. 1. Sensitivity of REM-2 type chamber filled with BF_3 up to 450 kPa, as a function of applied voltage. Hexagons – ^{252}Cf (neutrons alone, i.e. the photon component was corrected for), stars – $^{239}\text{Pu-Be}$ (neutrons alone), triangles – gamma radiation of ^{137}Cs source.

The chambers were used in a series of measurements in the treatment room, outside the irradiation field of 15 MV accelerator Varian Clinac 2300C/D photon beam, at the Oncology Centre in Warsaw.

It was shown, that there was no need to add a heavy moderator to the chamber – 1 kg of paraffin was sufficient in the case of large chamber REM-2 type. A simple free air ionisation chamber with electrodes covered by boron can also be used for monitoring of mixed radiation fields at medical accelerators.

There was no need to use boron enriched in ^{10}B in this work. However, the chambers filled with $^{10}\text{BF}_3$ can be also of interest, because they can be used as “photon insensitive” detectors e.g. in the measurements using two-detectors technique.

References

- [1] N. Golnik et al., in Proc. Tenth Symposium on Neutron Dosimetry, Uppsala, Sweden (2006)
- [2] N. Golnik et al., Raport IAE-126/A (2006)

RECOMBINATION PROCESSES IN HIGH-PRESSURE IONIZATION CHAMBERS IRRADIATED WITH LOW-LET RADIATION

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Experimental data obtained with recombination chambers, filled with different pressurized hydrocarbon gases [1], allowed for critical analysis of initial recombination, volume recombination and back diffusion of ions, being the main processes influencing the ion collection efficiency in the chambers. The new findings concern the dependence of volume recombination on gas density for different gases and the values of initial recombination for very low-LET radiation.

For steady radiation and given polarizing voltage U , the ion collection efficiency in the volume recombination process, f_v , can be expressed as:

$$f_v = (1 + a_v i / U^2)^{-1}$$

where a_v depends on the distance between electrodes and also on gas parameters, especially on the gas density. For given ionization chamber, a_v is constant.

The values of a_v were measured earlier [2] for the REM-2 chamber filled with methane or tissue equivalent gas at the gas pressure up to 1 MPa. In this work, the experimental studies were extended to the chambers containing different hydrocarbon gases and filled to higher gas pressures. It was shown, that a_v was proportional to the gas density and did not depend on the kind of the gas (Fig. 1).

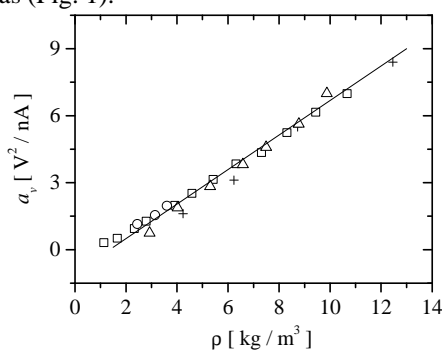


Fig. 1. Dependence of the coefficient a_v on the gas density for the REM-2 chamber (squares for methane, crosses for ethane, circles for ethylene and triangles for propane).

Initial recombination depends on local ion density within the tracks of charged particles in small volumes of traversed medium (with mass of order of one femtogram). The ion collection efficiency in the initial recombination process, f_i , was described by a formula:

$$f_i = (1 + m \mu)^{-1}$$

where μ is the relative local ion density, defined in such a way that $\mu = 1$ for gamma radiation of ^{137}Cs , and m is a function of polarizing voltage, U , and of the gas pressure, p . It's values can be determined experimentally as:

$$m = (1 - f_{Cs}) / f_{Cs}$$

where f_{Cs} is the ion collection efficiency at a definite voltage, measured in gamma radiation field of ^{137}Cs .

For radiations with LET higher than approximately 3.5 keV/ μm , μ can be correlated with unrestricted LET. It was expected [2], that for radiations with lower LET this correlation fails and the initial recombination decreases slower than it was theoretically predicted. In this work, the earlier reported [3] measurements of initial recombination in radiation fields of low-LET radiation were completed with the measurements using a ^{24}Na radiation source and accelerator beams of monoenergetic photons with energy of 6.62 MeV and with 9 MeV electrons from medical accelerator. The measurements were performed at several collecting voltages and the values of μ were determined from the plot $(1 - f_i) / f_i$ versus $(1 - f_{Cs}) / f_{Cs}$ [4]. For irradiation with ^{24}Na $\mu = 0.94 \pm 0.04$ was found. It is in good agreement with the ratio of restricted LET's $L_{500,D} / (L_{500,D})_{Cs}$, while the ratio of the corresponding values of unrestricted LET is of about 0.57.

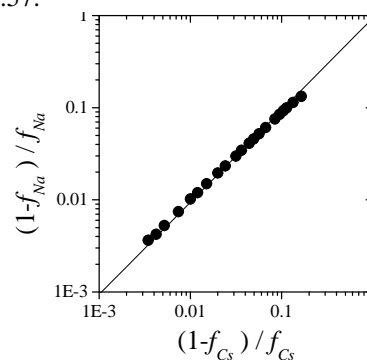


Fig. 2. Illustration of the method used for determination of local ionization density, μ for gamma radiation of ^{24}Na ($E = 2.75 \text{ MeV}$).

References

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SPECTROMETRIC MEASUREMENTS OF ^{131}I AND $^{99\text{m}}\text{Tc}$ ACTIVITY IN THYROID

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The aim of the research was to investigate a modified method for calibration of the thyroid counter with ^{131}I in order to take into account the depth of the thyroid in human neck [1]. It was shown earlier [2,3] that the cases of non-typical distribution of the iodine in the patient's neck could be detected by spectrometric measurements of the whole spectrum of the radiation emitted from the thyroid.

Another goal of this work was to improve the accuracy of the method by the use of HPGe counter instead of the NaI(Tl) scintillation detector and the changing the patient's position from sitting to lying.

The proposed calibration method was also studied for technetium $^{99\text{m}}\text{Tc}$, which is also used for diagnostic tests of thyroid.

The special phantoms (water and anatomic ones) were designed for this research.

Construction of the water phantom makes it possible to move the small vessels in two directions (horizontal and vertical).

The second phantom is an anatomically shaped chest with neck. The phantom is made with paraffin and contains human bones. The thyroid is simulated by the small vessels from the water phantom, which can be placed at various height.

The NaI detector was calibrated with two radioisotopes – iodine ^{131}I and technetium $^{99\text{m}}\text{Tc}$. The energy spectra of the radiation emitted from the phantoms were recorded for various simulated depths of the thyroid gland and various distance between the detector and the phantom surface.

The calibration of the thyroid counter with technetium $^{99\text{m}}\text{Tc}$ was performed in similar way as for ^{131}I . The same water phantom was used, with technetium solution in two vessels simulating thyroid.

The anatomic phantom was used for simulating thyroid diseases when the gland's position is different from the case of healthy patient. For each position, the parameter $S(h)$ – value of parameter S for different vertical position of thyroid was determined. Parameter S is the ratio of the counting rate recorded for the 364 keV

peak to the counting rate in two different Compton scattering bands – 100-150 keV for S_1 and 110-140 keV for S_2 . The ratio $S(h)/S(0)$ is shown in Fig. 1.

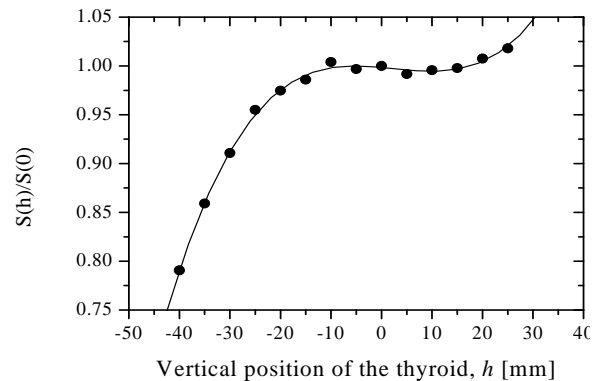


Fig. 1. The parameter S versus the vertical position of the thyroid phantom.

The results of the present studies confirm that the accuracy of the measurements of ^{131}I and $^{99\text{m}}\text{Tc}$ activity in thyroid gland can be considerably influenced by the change of the thyroid position. Especially large uncertainty can be caused by the displacement of the thyroid along its vertical axis and by partial shielding of the radiation sources by bones. Spectrometric measurements can be used in order to recognize the patients for whom the standard measurements give incorrect results.

The results of measurements with the HpGe detector show that proposed method can be used for detectors other than NaI.

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RECOMBINATION CHAMBERS FOR BORON NEUTRON CAPTURE THERAPY

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Dosimetric characterization of therapy beams for boron neutron capture therapy (BNCT) involves determination of four dose components: "boron" absorbed dose due to alpha and ${}^7\text{Li}$ charged particles, generated by the neutron capture reaction with ${}^{10}\text{B}$; gamma radiation component; fast neutron component and "nitrogen" dose due to protons generated by neutron capture on ${}^{14}\text{N}$. Microdosimetric characterization of the BNCT beam can be useful for monitoring of its quality and of possible variations of the beam composition in time. The possible detector is a tissue-equivalent recombination chamber. Parallel plate recombination chambers are known as reliable detectors for determination of gamma and high-LET dose components and for characterization of radiation quality of mixed radiation fields. Specially designed chambers can operate correctly even at high dose rates of therapeutic beams. However, parallel plate ionization chambers are less convenient for the routine dosimetry than cylindrical ones. Therefore, in this work [1,2] the measurements with a parallel-plate in-phantom chamber F1 were completed by investigations of a set of cylindrical chambers including a tissue equivalent (TE) chamber and three graphite chambers filled with different gases – CO_2 , N_2 and ${}^{10}\text{BF}_3$, in order to determine the thermal neutrons, ${}^{14}\text{N}$ capture, gamma, and fast neutron dose components. The separation of the dose components was based on differences of the shapes of the saturation curves, in dependence on LET spectrum of the investigated radiation.

The cylindrical chambers of T5 and G5 type form a set of pen-like ionization chambers, designed for determination of dose components at dose rates up to 500 Gy/h. The electrodes of T5 chamber are made with tissue-equivalent material. G5 is a graphite ionization chamber. The chambers were filled with different gases and marked with numbers indicated in Table 1.

Table 1. Parameters of cylindrical chambers.

Chamber	Gas	Gas pressure [MPa]	Sensitivity to ${}^{137}\text{Cs}$ radiation [pA/mGyh^{-1}]
T56	TE	0.50	0.3109
G51	N_2	2.00	1.4532
G58	CO_2	2.00	0.7800
G59	${}^{10}\text{BF}_3$	0.45	0.6175

The measurements were performed in the epithermal neutron beam at the BNCT facility [4] of LVR reactor of NRI Řež in Czech Republic. Saturation

curves were determined for F1, T56, G59 and G51 chambers in the reference point of the beam. The C- CO_2 chamber G58 operated in recombination mode at low polarizing voltage of 30 V. Under such conditions, the relative neutron sensitivity of the chamber was equal to 0.007, for the neutron spectrum of the BNCT beam.

The microdosimetric distribution of the absorbed dose versus restricted LET [3] was calculated from the measurements with F1 chamber for the following compartments of L_Δ : I – gamma radiation, II – 20–50 keV/ μm , III – 50–100 keV/ μm , IV – 100–200 keV/ μm . The number of the compartments was limited to four because the fitted values of the dose components for the compartments with higher LET were too small to be detected by this method. The values of total dose rate measured with the chamber, and of the dose components, derived to the experimental data are displayed in Table 2. The measurements with the cylindrical TE chamber T56 resulted in similar values of $D_\gamma = 218.1$ mGy/h and $D_{\text{total}} = 592$ mGy/h. The value of the gamma dose rate was measured also with G51 chamber resulting in the value of 219.4 mGy/h. In this case the gamma dose constituted 76.8% of the dose absorbed in nitrogen, when the chamber was irradiated free-in-air.

Table 2. Parameters of the BNCT beam measured with F1 chamber.

D_{total} [mGy/h]	D_γ [mGy/h]	$\frac{D_{20-50}}{D_{\text{total}}}$	$\frac{D_{50-100}}{D_{\text{total}}}$	$\frac{D_{100-200}}{D_{\text{total}}}$
597	220	0.12	0.31	0.16

The results of the measurements performed with the set of cylindrical chambers indicated that the recombination microdosimetric method can be used for characterization of BNCT beams. Similar results were obtained by parallel plate and cylindrical chambers.

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RADIONUCLIDE NEUTRON SOURCES IN CALIBRATION LABORATORY – NEUTRON AND GAMMA DOSES AND THEIR CHANGES IN TIME

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In the Institute of Atomic Energy (IAE) the standard fields of radionuclide neutron sources ²⁵²Cf, ²⁴¹Am-Be, as recommended by ISO [1], and of ²³⁹Pu-Be were established in late eighties [2], being traceable to National Physical Laboratory (NPL), Teddington, Great Britain. The parameters of neutron fields of these sources have been periodically measured, in order to check their stability. It was found that the change of neutron emission of the sources does not always follow the nominal decay of the main isotope.

The traceability of IAE sources to NPL has been renewed through the calibration of a transfer instrument, Studsvik Alnor 2202D neutron area dosimeter, in 1997 and in 2005, in standard fields of ²⁵²Cf and ²⁴¹Am-Be. The fluence response values were practically the same as those measured with IAE standard sources.

In 2006 the measurements of neutron and gamma components of ambient dose equivalent, $H^*(10)$, in 1 m distance from IAE neutron sources, have been performed. The following detectors have been used:

1. 2202D neutron area dosimeter, for determination of neutron emission rate and neutron ambient dose equivalent rate.
2. Hydrogen-free recombination chamber GW-2 for direct determination of gamma component of ambient dose equivalent.
3. Tissue equivalent ionisation chamber REM-2 for recombination microdosimetry measurements.

In Fig.1 the neutron emission of ²³⁹Pu-Be neutron source, growing up due to the admixtures [3], is shown.

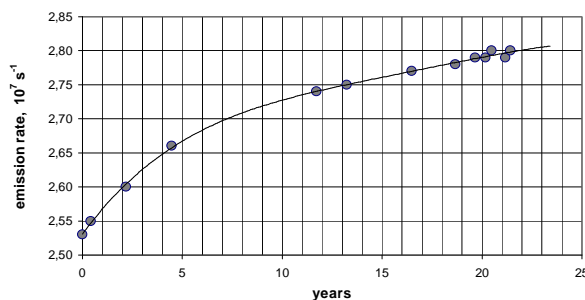


Fig. 1. ²³⁹Pu-Be neutron emission rate since October 26, 1984.

In Fig. 2 the ratio of gamma to neutron ambient dose equivalent is shown. It is stable for ²⁴¹Am-Be and ²³⁹Pu-Be sources but is growing up for old ²⁵²Cf source.

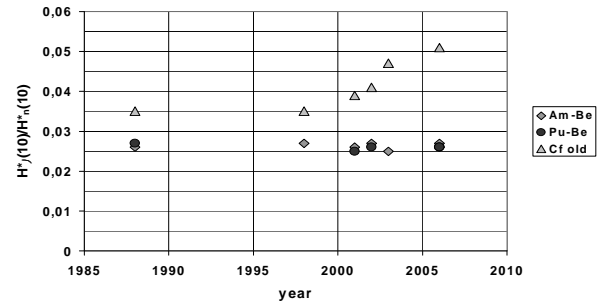


Fig. 2. Ratio of gamma to neutron dose equivalent rate.

Fig.3 presents the ratio of measured to calculated neutron emission rate of old ²⁵²Cf neutron source.

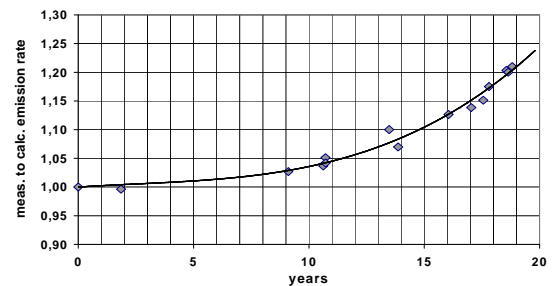


Fig. 3. Ratio of measured to calculated neutron emission of old ²⁵²Cf neutron source since 1987.

The results of measurements, collected in nearly twenty years, show that ²⁴¹Am-Be neutron source is really stable, both in neutron and gamma dose equivalent rates. For ²³⁹Pu-Be neutron source a correction for its growing up emission should be introduced; gamma ray contribution in total dose is stable and similar to that of ²⁴¹Am-Be source. ²⁵²Cf neutron sources decay quickly and not according to decay rate of the main radionuclide [4]. They are valuable for calibration laboratories because of pure fission neutron spectrum and great specific activity, but their parameters should be periodically checked.

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KEY COMPARISON FOR THE CALIBRATION OF AMBIENT DOSE EQUIVALENT METERS IN ISO NEUTRON REFERENCE FIELDS

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In the frames of EUROMET Project No 608 a key comparison for the calibration of ambient dose equivalent meters in ISO neutron reference fields has been organized. The purpose of the comparison is the determination of calibration factor $N_{H^*(10)}$ of transfer instrument N91:

$$N_{H^*(10)} = \frac{H^*(10)_n}{M_c}$$

where:

- $N_{H^*(10)}$ - calibration factor
 $H^*(10)_n$ - conventionally true value of neutron ambient dose equivalent rate
 M_c - measured value.

The transfer instrument consisted of Portable Neutron Monitor Model N91 (Harwell Instruments) [1]. The computerised measuring/recording system, elaborated and applied in the Laboratory, has been used in all experiments.

The sources employed in the Institute of Atomic Energy (IAE), recommended by ISO [2], were:

- $^{241}\text{Am-Be}$ neutron source AMN 24, nominal activity ca. 185 GBq, $T_{1/2} = 432$ years. Total emission, determined in manganese bath in National Physical Laboratory, Teddington, UK, was equal to $(1.138 \pm 0.10) \times 10^7 \text{ s}^{-1}$ on 14 June 1984. Anisotropy factor equal to 1.035, determined experimentally in IAE laboratory [3]. Scattered neutrons component measured with shadow cone.
- ^{252}Cf neutron source, nominal 100 μg (2 GBq) ^{252}Cf (spontaneous fission), $T_{1/2} = 2.65$ years. Total emission, determined in manganese bath in National Physical Laboratory, Teddington, UK, was equal to $(2.325 \pm 0.023) \times 10^8 \text{ s}^{-1}$ on 2 March 2005. Anisotropy factor equal to 1.012, according to Hunt [4]. Scattered neutrons component measured with shadow cone.

Calibration room in IAE, has dimensions ca. 4 m \times 4 m \times 16 m. Neutron sources are stored in iron plugs inside concrete channels on the depth of 1.3 m. For the experiments or calibrations, a neutron source can be exposed on the light aluminium stand, 102 cm over the floor. The instrument under examination is placed on a support or a rack mounted on a trolley, which can be

moved along calibration hall. The main calibration position is at the distance of 100 cm from the source centre.

Amongst the methods of calibration, described in ISO documents [5, 6], the calibration with shadow cone is the most suitable for IAE calibration hall.

The neutron fields in IAE are traceable to National Physical Laboratory (NPL). Verification of the neutron sources emission through recalibration of a transfer instrument 2202D neutron remmeter has been done in 1997 and 2005 at NPL with standard sources $^{241}\text{Am-Be}$ and ^{252}Cf .

Two methods were applied for the determination of calibration factors $N_{H^*(10)}$:

Method 1: traceable to source calibration – involved the calculation of neutron flux density (fluence rate) in a given position, calculation of neutron ambient dose equivalent rate, calculation of calibration factor – a ratio of conventionally true value of $H^*(10)$ to number of counts, N , applying relevant corrections.

Method 2: traceable to instrument calibration – comparison of counting rates of 2202D and N91; calculation of calibration factor for N91 in comparison with that of 2202D, determined at NPL as follows:

- for $^{241}\text{Am-Be}$: 0.875 nSv/pulse ($\pm 1.5\%$);
 for ^{252}Cf : 0.824 nSv/pulse ($\pm 1.7\%$).

Table 1. Results of $N_{H^*(10)}$ determination for N91.

Source	Method 1	Method 2
$^{241}\text{Am-Be}$	(1.81 ± 0.16) nSv/pulse	(1.81 ± 0.08) nSv/pulse
^{252}Cf	(1.55 ± 0.06) nSv/pulse	$1.58 \pm 0.07)$ nSv/pulse

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